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Part V

Nuclear Regulatory Commission

Domestic Licensing of Production and Utilization Facilities; Fracture Toughness Requirements for Nuclear Power Reactors

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Domestic Licensing of Production and Utilization Facilities; Fracture Toughness Requirements for Nuclear Power Reactors

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is considering amending its regulations which specify fracture toughness requirements for nuclear power reactors and its requirements for reactor vessel material surveillance programs. The amendments would clarify the applicability of these requirements to old and new plants, modify certain requirements as described below, and shorten and simplify these regulations by more extensively incorporating by reference appropriate National Standards.

DATES: Comment period expires January 13, 1981.

ADDRESSES: Written comments should be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

FOR FURTHER INFORMATION CONTACT: Dr. P. N. Randall, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, 301–443–5997.

SUPPLEMENTARY INFORMATION: When 10 CFR Part 50 was amended by adding Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," on July 17, 1973 (38 FR 19012), the ASME Boiler and Pressure Vessel Code (the ASME Code) provisions pertaining to nuclear power plant components were being extensively revised. Some requirements that had not yet become effective in the ASME Code were included in Appendix G. Now that experience has shown that the ASME Code requirements are adequate, Appendix G is being condensed by more extensively incorporating the ASME Code by reference.

Appendix H incorporated by reference the 1973 edition of ASTM E 185.[1] Now there is a new edition, E 185–79. As amended, Appendix H would specify the earliest edition of E 185 that could be used for each part of the surveillance program, and would delete paragraphs that are now covered by specific provisions of E 185–79.

Paragraph 50.55a(i), which makes the provisions of Appendices G and H conditions of a construction permit for a utilization facility (e.g., a nuclear power plant), would be deleted and a new § 50.47 would be added for this purpose. This change would avoid any confusion that might occur when § 50.55a is amended for purposes not relevant to the appendices. The new § 50.47 also would describe when an exemption must be requested of the Commission for acceptance of a proposed alternative to the requirements of Appendices G and H. Conversely, several paragraphs of Appendices G and H would be amended to state that a proposed alternative may be accepted by the Director of Nuclear Reactor Regulation if the proposed alternative provides a margin of safety equivalent to that achieved by the specified requirement.

Significant differences between the existing regulations in Appendix G and the proposed amendments are: [2]

¶II.G. The definition of "adjusted reference temperature" would be changed to make the definition applicable to test data over a broader range of severity of radiation damage.

¶II.H. The definition of "beltline" would be changed to more clearly delineate which parts of the vessel are of concern from the standpoint of surveillance of radiation damage.

¶II.J. The definition of "integrated surveillance program" would be removed from Appendix G and a more complete discussion of such a program would be included in paragraph II.C. of Appendix H.

§ III. Section III, "Fracture Toughness Tests," would be reduced in length because most of the pertinent requirements could be incorporated by referencing the ASME Code.

¶III.A. Language would be added to paragraph III.A. to clarify how the fracture toughness test requirements would be applied to "old" plants, those for which the reactor vessel was constructed to an ASME Code earlier than the Summer 1972 Addenda to the 1971 Edition. This language would clarify the Commission's intention that all plants must meet the required margins of safety and the other fracture prevention requirements of Appendices G and H; but the owners of "old" plants may, when approved by the Director of Nuclear Reactor Regulation, use fracture toughness tests and fracture analyses other than those required by Appendices G and H to demonstrate that the required margins of safety have been met.

§ IV. As amended, Section IV, "Fracture Toughness Requirements" would incorporate by reference the fracture toughness requirements of the ASME Code and then give certain supplemental requirements, thereby deleting much technical detail from the regulation.

¶IV.A.2.b. As an alternative to the existing general requirement for fracture toughness of nozzles, flanges and shell regions near structural discontinuities, a specific pressure-temperature requirement would be added.

¶IV.A.2.c. For boiling water reactors, the criticality limit would be reduced to permit them to go critical earlier during startup. A systems analysis has shown that the existing limit is more restrictive than is necessary to provide the required safety margin.

¶IV.A.2.d. The minimum permissible temperature for system hydrostatic pressure tests would be reduced when performed when there is no fuel in the reactor (and thus no radiological hazard to the public) to improve the efficiency of the inspection operation.

¶IV.B. The 50 ft lb requirement for Charpy upper-shelf energy would be stated explicitly. Language would be added to permit acceptance of a lower value by the Director of Nuclear Reactor Regulation if a fracture analysis showed that the margin of safety against ductile fracture was equivalent to the margin of safety against fracture in the transition region required by Appendix G of the ASME Code.

¶V.B. A requirement would be added that material toughness values to be used in fracture analyses are those predicted for the material near the tip of the assumed flaw at its deepest part.

Significant differences between the existing regulations in Apendix H and the proposed amendments are:

¶II.B. As revised, paragraph II.B. would incorporate by reference ASTM E 185, and reference to the 1973 edition of ASTM E 185 would be deleted. Language would be added to permit the use of the 1979 edition or an earlier edition if it was the current edition at the time the action was taken. For example, the earliest

[1] Standard Recommended Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H St., NW., Washington, D.C.

[2] Listed according to existing paragraph numbers. See Enclosure 3.

edition of ASTM E 185 that could be used in the selection of surveillance materials, preparation of specimens, and construction of surveillance capsules would be the edition that was current on the issue date of the ASME Code to which the reactor vessel was purchased.

¶II.C.3. Most of this paragraph would be deleted, because the requirements for withdrawal schedules contained in the 1979 edition of ASTM E 185 provide satisfactory criteria for scheduling surveillance information gathering.

¶II.C.4. Expanded criteria would be given for an integrated surveillance program for a set of reactors of similar design and operating features. In an integrated program, surveillance capsules from one or more reactors are irradiated in other reactors in the set.

§III. Section III would be deleted because its provisions are covered in Appendix G, Section V.

§IV. Reporting requirements for the test results from each capsule withdrawal would be clarified and a time limit for submittal of the report would be specified.

Staff of the Commission's Office of Standards Development has prepared a value/impact statement for the proposed amendment, which provides additional technical details and justification. This statement is available for inspection by the public in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C. Single copies of the value/impact statement may be obtained by request addressed to the Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: P. N. Randall, (301) 443–5997.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganizaiton Act of 1974, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments of 10 CFR Part 50 is contemplated. All interested persons who wish to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch by Janaury 13, 1981. Copies of comments received may be examined in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C.

1. Section 50.55a is amended by deleting paragraph (i) and inserting the word "Reserved" in its place.

2. A new § 50.47 is added to 10 CFR Part 50 to read as follows:

§ 50.47 Acceptance criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors.

(a)(1) Except as provided in paragraph (a)(2) of this section all light-water nuclear power reactors shall meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in Appendices G and H to this part.

(2) Proposed alternatives to the described requirements or portions therof may be used when an exemption is granted by the Commission as authorized by § 50.12. In addition, the applicant must demonstrate that (i) compliance with the specified requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety and (ii) the proposed alternatives would provide an adequate level of quality and safety.

(b) [Reserved]

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§ 50.12 [Amended]

3. Paragraph (a) of § 50.12 is amended by adding the following sentence at the end of the paragraph:

* * * To obtain an exemption to Appendices G and H to this part, the requirements of paragraph 50.47(a)(2) must be met in addition to the requirements of this paragraph.

4. Appendices G and H to 10 CFR Part 50 are revised to read as follows:

Appendix G—Fracture Toughness Requirements

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I. Introduction and Scope
II. Definitions
III. Fracture Toughness Tests
IV. Fracture Toughness Requirements
V. Inservice Requirements—Reactor Vessel Beltline Materials

I. Introduction and Scope

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G–2110 of the ASME Code.[1]

B. Welds and weld heat-affected zones in the materials specified in section I.A.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

Adequacy of the fracture toughness of other ferritic materials shall be demonstrated to the Director of Nuclear Reactor Regulation on an individual case basis.

II. Definitions

A. "ASME Code" means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components." "Section XI" means Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components." If no edition or addenda is specified, the applicable ASME Code edition and addenda [2] and any limitations and modifications thereof are specified by § 50.55a, Codes and Standards.

B. "Ferritic material" means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

C. "System hydrostatic tests" means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

D. "Specified minimum yield strength" means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built pursuant to § 50.55a.

E. "Reference temperature" means the reference temperature, RT_{NDT}, as defined in the ASME Code.

F. "Adjusted reference temperature" means the reference temperature as adjusted for irradiation effects (see Section V of this Appendix) by adding to RT_{NDT} the temperature shift in the average Charpy curve for the irradiated material relative to that for the unirradiated material, measured at the 30 ft lb (41J) level.

G. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

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[1] Defined in paragraph II.A. The latest Edition and Addenda permitted by paragraph 50.55a(b) at the time the analysis is made shall be used for the purposes of paragraph I.A.

[2] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. Copies are available for inspection at the Commission's Public Document Room, 1717 H St. N.W., Washington, D.C.

III. Fracture Toughness Tests

A. To demonstrate compliance with the fracture toughness requirements of Sections IV and V of this appendix, ferritic materials shall be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of Appendix H. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (pursuant to § 50.55a), the fracture toughness data and data analyses shall be supplemented in a manner approved by the Director of Nuclear Reactor Regulation to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph V.C.2 shall be submitted to and approved by the Director of Nuclear Reactor Regulation prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs A and B of this Section shall comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. Fracture Toughness Requirements

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials shall meet the requirements of the ASME Code supplemented as follows for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrances.

1. Reactor vessel beltline materials shall have minimum Charpy upper-shelf energy of 75 ft lb (102J) initially and shall maintain minimum upper-shelf energy throughout the life of the vessel of 50 ft lb (68J), unless it is demonstrated in a manner approved by the Director of Nuclear Reactor Regulation that lower values of upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of the ASME Code.[3]

2. When the core is not critical, pressure-temperature limits for the reactor vessel shall be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code [3] supplemented by the requirements of Section V of this Appendix. In addition, when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the stressed regions of nozzles, flanges and other structural discontinuities shall be at least 150° F above (83 C above) the reference temperature of the material in those regions unless a lower temperature can be justified by showing that the margins of safety for those regions are equivalent to those required for the beltline when it is controlling. The justification submitted for the pressure temperature limits shall describe the methods of analysis used.

3. When the core is critical (other than for the purpose of low-level physics tests) the temperature of the reactor vessel shall not be lower than 40° F above (22 C above) the minimum permissible temperature of paragraph 2. of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure, in which case the minimum permissible temperature is 60° F above (33 C above) the reference temperature of the closure flange regions that are highly stressed by the bolt preload.

4. If there is no fuel in the reactor during system hydrostatic pressure tests or leak tests, the minimum permissible test temperature shall be $RT_{NDT}+60°$ F ($RT_{NDT}+33$ C).

5. If there is fuel in the reactor during system pressure tests or leak tests, the requirements of paragraphs 2 or 3 shall apply, depending on whether the core is critical during the test.

B. Reactor vessels for which the predicted value of upper shelf-shelf energy at end of life is below 50 ft lb or the predicted value of adjusted reference temperature at end of life exceeds 200° F (93 C) shall be designed to permit a thermal annealing treatment to recover material toughness properties of ferritic materials of the reactor vessel beltline.

V. Inservice Requirements—Reactor Vessel Beltline Material

A. The effects of neutron radiation on the reference temperature and upper-shelf energy of reactor vessel beltline materials, including welds, shall be predicted from the results of pertinent radiation effects studies in addition to the results of the surveillance program of Appendix H.

B. Reactor vessels may continue to be operated only for that service period within which the requirements of Section IV are satisfied using the predicted value of the adjusted reference temperature and the predicted value of the upper-shelf energy at the end of the service period to account for the effects of radiation on the fracture toughness of the beltline materials. These predictions shall be made for the radiation conditions at the tip of the assumed flaw at its deepest part. The highest adjusted reference temperature and the lowest upper-shelf energy level of all the beltline materials shall be used to verify that the fracture toughness requirements are satisfied.

C. In the event that the requirements of Section V.B. cannot be satisfied, reactor vessels may continue to be operated provided all of the following requirements are satisfied:

1. A volumetric examination of the beltline materials that do not satisfy the requirements of Section V.B. including 100 percent of any associated welds shall be made and any flaws evaluated according to Section XI of the ASME Code and as otherwise specified by the Director of Nuclear Reactor Regulation.

2. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation shall be obtained from results of supplemental fracture toughness tests.

3. An analysis shall be performed that conservatively demonstrates, making appropriate allowances for all uncertainties, the existence of equivalent margins of safety for continued operation.

D. If the procedures of section V.C. do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may, subject to the approval of the Director of Nuclear Reactor Regulation, be given a thermal annealing treatment to recover the fracture toughness of the material. The degree of recovery shall be measured by testing additional specimens that have been withdrawn from the surveillance program capsules and that have been annealed under the same time-at-temperature conditions as those given the beltline material. The results shall provide the basis for establishing the adjusted reference temperature and upper shelf energy after annealing. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. using the values of adjusted reference temperature and upper-shelf energy that include the effects of annealing and subsequent irradiation.

E. The proposed programs for satisfying the requirements of Sections V.C. and V.D. shall be reported to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, for review and approval on an individual case basis at least 3 years prior to the date when the predicted fracture toughness levels will no longer satisfy the requirements of Section V.B.

Appendix H—Reactor Vessel Material Surveillance Program Requirements

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I. Introduction
II. Surveillance Program Criteria
III. Report of Test Results

I. Introduction

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from their exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in Sections IV and V of Appendix G.

II. Surveillance Program Criteria

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E<1MeV) at the end of the design life of the vessel will not exceed 10^{17} n/cm^2.

[3] The latest Edition and Addenda permitted by paragraph 50.55a(b) at the time the analysis is made shall be used for the purposes of paragraphs IV.A.1 and IV.A.2.

B. Reactor vessels that do not meet the conditions of paragraph II.A. shall have their beltline materials monitored by a surveillance program complying with ASTM E 185,[1] except as modified by this Appendix. ASTM E 185 was approved for incorporation by reference by the Director of the Federal Register, on May 29, 1973.

1. That part of the surveillance program conducted prior to the first capsule withdrawal shall meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. For each capsule withdrawal, the test procedures and reporting requirements shall meet the requirements of the edition of E 185 in effect on the date of capsule withdrawal, to the extent practical for the configuration of the specimens in the capsule. For any part of the surveillance program, later editions of E 185 may be used instead of the editions previously specified, but including only those editions through 1979.

2. Surveillance specimen capsules shall be located near the inside vessel wall in the beltline region, so that the specimen irradiation history duplicates to the extent practicable, within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds shall be done according to the requirements for permanent structural attachments to reactor vessels given in the ASME Code, Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in ASTM E 185.

3. Proposed withdrawal schedules shall be submitted with a technical justification therefor to the Director of Nuclear Reactor Regulation for approval. The proposed schedule shall not be implemented without prior approval.

C. An integrated surveillance program may be considered for a set of reactors that have similar design and operating features. The representative materials chosen for surveillance from each reactor in the set may be irradiated in one or more of the reactors, but there must be an adequate dosimetry program for each reactor. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted, but the amount of testing may be reduced if the initial results agree with predictions. Integrated surveillance programs must be approved by the Director of Nuclear Reactor Regulation on a case-by-case basis. Criteria for approval include the following considerations:

1. The design and operating features of the reactors in the set shall be sufficiently similar to permit accurate comparisons of the predicted amount of radiation damage as a function of total power output.

2. There shall be adequate arrangements for data sharing between plants.

3. There shall be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

4. There shall be substantial advantages to be gained in reduced power outages or personnel exposure to radiation, achieved by not requiring surveillance capsules in all reactors in the set.

III. Report of Test Results

A. Each capsule withdrawal and the results of the fracture toughness tests shall be the subject of a summary technical report to be submitted for approval to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 90 days after completion of testing. The Director shall be notified at least 30 days in advance of the capsule withdrawal, giving the expected date of completion of testing and submittal of report.

B. The report shall include the data required by ASTM E 185, as required by paragraph II.B.1, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the operating pressure-temperature limits given in the Technical Specifications is required, the revised limits shall be submitted with the report, including any changes made in operating procedures required to meet the limits.

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(Secs. 103, 104, 161i, Pub. L. 83–703; 68 Stat. 936, 937, 948; Sec. 201, Pub. L. 93–438, 88 Stat. 1242; (42 U.S.C. 2133, 2134, 2201(i), 5841))

Dated at Washington, D.C. this 5th day of November 1980.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 80-35207 Filed 11-13-80; 8:45 am]

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[1] Standard Recommended Practice for Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels. Copies may be obtained from the American Society for Testing and Materials, 1916 Race St., Philadelphia, Pa. 19103. Copies will be available for inspection at the Commission's Public Document Room, 1717 H St., NW., Washington, D.C.